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February 6, 1997
PY-CEI/NRR-2134L

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 97-001

Ladies and Gentlemen:

Enclosed is Licensee Event Report (LER) 97-001, Nonlicensed Operator Electrical Switching Error Results in Reactor Protection System and Other Engineered Safety Feature Actuations.

If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Manager - Regulatory Affairs, at (216) 280-5606.

Very truly yours,

for Lew. W. Myers

Enclosure: LER 97-001

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager

CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9702100405 DOC.DATE: 97/02/06 NOTARIZED: NO DOCKET #
FACIL:50-440 Perry Nuclear Power Plant, Unit 1, Cleveland Electric 05000440
AUTH.NAME AUTHOR AFFILIATION
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MYERS,L.W. Centerior Energy
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-001-00 on 970107,nonlicensed operator electrical switching error results in reactor protection sys & other engineered safety feature actuations occurred.Caused by personnel error.Nonlicensed were counseled.W/970206 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:Application for permit renewal filed. 05000440

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LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4)
Nonlicensed Operator Electrical Switching Error Results in Reactor Protection System and Other Engineered Safety Feature Actuations.

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	07	97	97 --	001 --	00	02	06	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)	X	OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Keith R. Jury, Supervisor-Compliance	(216) 280-5594

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 7, 1997, at 0533 hours, the Perry Nuclear Power Plant, Unit 1, automatically scrammed from 100 percent rated thermal power, due to a loss of electrical power to the Feedwater Control system which resulted in a Reactor Protection system actuation on low reactor water level. The High Pressure Core Spray system, the Division 3 Diesel Generator, and associated support systems including Division 3 Emergency Service Water (ESW), the Reactor Core Isolation Cooling system, the Division 1 ESW system, and the Nuclear Steam Supply Shutoff system also actuated during the event.

The cause of this event is personnel error; a Nonlicensed Operator failed to utilize self-checking and performed an inappropriate procedural step which resulted in the loss of electrical power to the vital loads. The pre-job briefing conducted for the switching activity did not contain sufficient detail to ensure successful completion of the switching activity.

A Standing Instruction was issued to inform the operating crews of the event, to communicate expectations concerning in-field supervision and pre-job briefings, and to detail missed opportunities for preventing the event through self-checking and checking of others. Procedural enhancements are being evaluated and programmatic improvements are being developed.

This event is being reported in accordance with the requirements of 10CFR50.73(a)(2)(iv). This report is also being submitted to fulfill the requirements of Operational Requirements Manual, Special Reports, Emergency Core Cooling System Injection, Section 7.6.2.1.

I. Introduction

On January 7, 1997, at 0533 hours, the Perry Nuclear Power Plant (PNPP), Unit 1, automatically scrammed from 100 percent rated thermal power, due to a loss of electrical power to the Feedwater Control system [SJ] which resulted in a Reactor Protection system [JC] (RPS) actuation on low reactor pressure vessel (RPV) water level. Additionally, the High Pressure Core Spray (HPCS) system [BG], Division 3 Diesel Generator [EK] and associated support systems including the Division 3 Emergency Service Water [BI] (ESW), the Reactor Core Isolation Cooling (RCIC) system [BN], the Division 1 ESW system, and the Nuclear Steam Supply Shutoff system [JM] (NSSSS) actuated. During the transient, RPV water level decreased to approximately 83.5 inches above the top of active fuel. RPV water level was restored utilizing the HPCS and RCIC systems.

Notification was made to the NRC via the Emergency Notification System at 0615 hours (ENF No. 31549), in accordance with the requirements of 10CFR50.72 (b)(1)(iv) as an event that resulted in Emergency Core Cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal, and 10CFR50.72(b)(2)(ii), as an event that resulted in an automatic actuation of any engineered safety feature (ESF), which in this case includes the RPS, HPCS system and its associated auxiliaries, ESW system, and NSSSS. This event is being reported in accordance with 10CFR50.73(a)(2)(iv) for an event that resulted in ESF actuations.

Submittal of this report also satisfies the Operational Requirements Manual, Special Reports, ECCS Injection, Section 7.6.2.1, which requires a special report following any ECCS actuation and injection into the Reactor Coolant system. This was the eleventh HPCS injection to date. The injection nozzle usage factor is currently less than 0.70.

At the time of the event, the plant was in Mode 1 at 100 percent of rated thermal power. The RPV pressure was at approximately 1025 psig with reactor coolant at saturated conditions.

II. Event Description

On January 6, 1997, at approximately 1909 hours, the operations crew involved with this event assumed shift duties. One task assigned to the shift was to perform an evolution in support of the next day's scheduled work activities. This evolution was to shift Non-Essential Vital Power Supply DB-1-A [EE] to its alternate source and to shutdown the DB-1-A inverter in support of opening a roll-up door to move a large piece of equipment from outdoors into the Turbine Power Complex. Previous experience had demonstrated that opening the Turbine Power Complex roll-up door with outside air temperature below 40 degrees Fahrenheit (F) could result in tripping of the DB-1-A inverter, and the removal of the inverter from service was procedurally accommodated under these conditions by System Operating Instruction (SOI)-R14, "120V AC Vital Inverters (Unit 1)". All Emergency Core Cooling systems and the associated diesel generators were operable and in standby.

The Senior Reactor Operator (SRO) and Operations Foreman (a licensed reactor operator normally assigned to supervise Nonlicensed Operator activities) discussed plans for the power switching task to take place later in the shift and the need to provide direct field supervision. Both the SRO and the Operations Foreman agreed that the Operations Foreman should remain in the control room. The Operations Foreman assigned the switching task to a Nonlicensed Operator and held a pre-job briefing to discuss the job assignment. The briefing included discussion of the sections of SOI-R14 to be performed.

At approximately 0500 hours, the Nonlicensed Operator reviewed SOI-R14 and then proceeded to the turbine power complex to begin the switching task. Although the Nonlicensed Operator maintained contact with control room personnel during the switching task, at approximately 0533 hours, he inappropriately performed steps in SOI-R14 which resulted in loss of electrical power to various plant systems and indications. The equipment/systems affected by this power loss included process computer screen displays and input modules, the Post Accident Sampling System, the full core display for the Rod Control and Information system, reactor level control, and the speed controllers for the reactor feed pump turbines. The loss of electrical power to reactor level control and speed controllers resulted in loss of reactor feedwater and the downshifting to slow speed of the Reactor Recirculation pumps.

At approximately 0534 hours, RPV water level decreased to Level 3 (178 inches above the top of active fuel (TAF)) and the RPS initiated, causing a reactor scram. RPV water level continued to decrease to Level 2 (130 inches above the TAF), which resulted in the following:

The HPCS system, the Division 3 Diesel Generator and associated support systems including Division 3 Emergency Service Water (ESW), the RCIC system, and the Division 1 ESW system automatically initiated.

The Main Turbine and Reactor Feedpump turbines tripped; the Main Generator tripped on reverse power.

The NSSSS actuations/isolations occurred, including isolation of the Reactor Water Cleanup system.

The Motor-driven Feedwater Pump automatically started.

The Main Turbine Bypass valves opened and closed as designed to control pressure.

The Redundant Reactivity Control system (RRCS) actuated to trip the Reactor Recirculation pumps.

The reactor mode switch was subsequently placed in the "Shutdown" position and RPV water level continued to decrease to a minimum level of approximately 83.5 inches above TAF before injection of the HPCS and RCIC systems began to increase RPV water level. Plant Emergency Instruction (PEI)-B13, "Reactor Pressure Vessel Control," was entered due to the Level 3 scram. The total time that RPV water level was less than Level 3 during the transient was slightly less than two minutes and twenty-five seconds. Normal control rod position indication at the reactor control front panel was not available due to the DB-1-A switching error. All control rods were verified to be fully inserted into the core by use of alternate indications in the back panels of the control room.

At approximately 0536 hours, RPV water level increased above Level 2, and it was concurrently determined that the event was caused by the Nonlicensed Operator's switching activities on DB-1-A. At approximately 0539 hours, RPV water level had increased above Level 8 (219 inches above the TAF) which resulted in the HPCS and RCIC systems terminating injection and the Motor-driven Feedwater Pump tripping. At approximately 0542 hours, operations personnel began restoration of the electrical loads associated with DB-1-A in accordance with Off-Normal Instruction (ONI)-R25-2, "Loss of Non-Essential 120V Bus (Unit 1)," and other applicable procedures. The minimum RPV pressure for the transient (i.e., 620 psig) was reached at this time.

At approximately 0544 hours, suppression pool level was observed to be less than 17.8 feet. The SRO directed entry into PEI-T23, "Containment Control," and the RCIC system suction was realigned from the suppression pool to the condensate storage tank. At 0545 hours, the full core display on the reactor control front panel was restored and control rods were re-verified to be fully inserted into the core utilizing the display. A maximum RPV level of 240 inches above the TAF was reached at 0550 hours. By 0608 hours, the electrical loads associated with DB-1-A were fully restored. ONI-R25-2 was exited, ONI-B21-4, "Isolation Restoration (Unit 1)," was then entered to restore from NSSSS isolations and ONI-C71-1 was entered to begin the scram recovery. At 0620 hours, the Motor-driven Feedwater Pump was started and utilized to maintain RPV level between 185 and 215 inches above the TAF with RPV pressure between 800 and 1000 psig. HPCS was placed into "Standby Readiness" at 0633 hours.

By 0745 hours, RPS scram signals and RRCS signals were reset, and at approximately 0750 hours, while monitoring RPV cooldown rate, it was discovered that the 100 degree F/hour rate limit as well as the pressure/temperature limits checked by Technical Specification (TS) Surveillance Requirements (SRs) 3.4.11.1.b. and 3.4.11.1.a. were being exceeded by plant conditions. The TS Limiting Condition for Operation (LCO) 3.4.11 Actions were entered and performed as required. At 0804 hours PEI-T23 and PEI-B13 were exited. At 0835 hours, the Division 3 diesel generator was returned to "Standby Readiness", and at 0950 hours, the RCIC system was returned to standby readiness.

At 1110 hours, reactor engineering personnel reviewing the scram timing data identified three control rods that were "slow," however, the rods were operable in accordance with the requirements of TS LCO 3.1.4. The scram pilot solenoid valves were subsequently replaced on the hydraulic control units for the three "slow" control rods; the control rods were retested satisfactorily.

On January 8, 1997, at 1318 hours, the Reactor Coolant system was found to be acceptable for continued operation as required by TS LCO 3.4.11 ACTION a.2. The plant remained in Mode 3 until the plant entered Mode 2 on January 9, 1997, at 0337 hours and commenced reactor startup at 0405 hours.

III. Cause of Event

The cause of this event is personnel error; the Nonlicensed Operator failed to utilize self-checking and performed an inappropriate procedural step which resulted in the loss of electrical power to the vital loads. The pre-job briefing conducted for the DB-1-A switching activity did not contain sufficient detail to ensure its successful completion. The SRO assigned the Operations Foreman to develop the plan for the DB-1-A vital loads shift from the inverter to the alternate power supply and for the shutdown of the inverter. Both individuals reviewed SOI-R14 and independently developed an appropriate sequence for this evolution. The SRO and Operations Foreman compared their plans and concurred in the SOI-R14 sections and steps to be performed. Both individuals recognized that conditional steps to remove power from loads in the inverter shutdown section of the procedure should not be performed, but this was not verbalized during their discussion. When the Operations Foreman conducted a pre-job briefing with the Nonlicensed Operator, the briefing content omitted discussion of the specific procedural steps and that the conditional steps in the inverter shutdown section did not apply and should not be performed.

The Nonlicensed Operator reviewed the applicable sections of SOI-R14, as determined in the pre-job briefing, and developed his plan for performing the evolution. His plan included performance of the inappropriate steps listed above. The Operations Section has a verbal policy that requires supervision in the field for electrical switching evolutions; however, the policy was not consistently known or understood throughout the operating crews. The SRO allowed the Nonlicensed Operator to perform the switching activities without direct supervision. The Nonlicensed Operator left the control room to perform the switching activity with a preconceived (incorrect) idea of which steps of SOI-R14 were to be performed. These steps should have been thoroughly discussed during the pre-job briefing. Additionally, during the performance of the steps in the SOI, there were opportunities for the Nonlicensed Operator, as well as others, to detect the error through self-checking and checking of others, but these opportunities were missed.

IV. Safety Analysis

The Updated Safety Analysis Report (USAR) section 15.2.7, "Loss of Feedwater Flow," assumes a total loss of feedwater at high power with no change in reactor recirculation flow until Level 3 is reached (no Flow Control Valve Runback) and no HPCS nor RCIC system flow until 50 seconds into the transient. In this event, reactor recirculation pumps immediately downshifted reducing reactor power. RCIC and HPCS flow began approximately 27 seconds into the transient. The USAR

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

assumes the Main Steamline Isolation Valves (MSIVs) would close and that RCIC and Residual Heat Removal System would be used to maintain level and cool off the reactor. In this event, the MSIVs remained open, level control was with the Motor Feedwater Pump, and pressure control was maintained utilizing the main condenser. Operations successfully completed all applicable actions identified in USAR section 15.2.7.2.1.1 "Identification of Operator Actions," which include:

verification of all rods in, following the scram;

verification that the reactor recirculation pumps trip on reactor low level (Level 2);

verification of HPCS and RCIC initiations;

securing of HPCS when reactor level and pressure are being maintained;

monitoring of turbine coastdown and turbine auxiliaries; and,

completion of scram report and survey of maintenance requirements.

This event is bounded within the USAR analyses and is considered to have had minimal safety significance.

For the evaluation of the TS 3.4.11 Reactor Coolant system cooldown rate and pressure/temperature issues, PNPP and General Electric engineering staffs evaluated the transient sequence and data plots from process recorders and the Emergency Response Information system computer. The evaluation included a review of the fatigue usage for the recirculation loops (including the piping, valves, and pump) and reactor pressure vessel using a conservative bounding analytical transient of 535 degrees F to 120 degrees F cooldown in 3 minutes and 120 degrees F to 535 degrees F heatup in 3 minutes. This temperature range and time duration were utilized to simplify the evaluation and to ensure conservatism. A fracture mechanics evaluation of the reactor pressure vessel beltline and non-beltline regions was performed. Based on the evaluations, reasonable assurance was obtained that there was minimal impact on the reactor vessel and recirculation piping and that there was minimal impact on component stresses or cumulative fatigue limits due to transients after the scram. A multidisciplined team was assigned to investigate and address personnel performance, procedural, operational, and engineering aspects of the cooldown rate issue.

V. Similar Events

LERs 87-072, 88-012, 90-001 and 95-007 document other personnel error/feedwater induced reactor scram events.

LER 87-072 documented an event in which a licensed operator inadvertently de-energized the Hot Surge Tank low level trip logic during the transfer of station loads resulting in loss of feedwater flow to the reactor vessel with a subsequent reactor scram due to low reactor water level. The event was attributed to personnel error; inattention to detail on the part of the operator performing the switching evolution. Corrective actions included counseling of the individual and a design change to prevent a loss of power to the control logic from initiating a complete loss of feedwater control.

LER 88-012 documented an event in which an improper DC bus transfer performed by Nonlicensed Operators resulted in a loss of feedwater flow with a subsequent scram due to low reactor water level. The event was attributed to personnel error and inadequate procedure. Corrective actions included counseling of the individuals and revision of the applicable procedure.

LER 90-001 documented an event in which a Nonlicensed Operator inadvertently removed some wrong fuses as part of a 480 volt AC electrical switching evolution, which resulted in a momentary power interruption to the feedwater control circuitry and loss of feedwater flow with subsequent reactor scram due to low reactor water level. The event was attributed to personnel error (i.e., inattention to detail). Corrective actions included counseling of the Nonlicensed Operator, an SOI revision, production and use of a training video tape on the event, and a design change to increase reliability of feedwater control system electrical power during switching operations.

LER 95-007 documented an event in which a licensed operator error during a transfer of feedwater level control resulted in loss of feedwater flow and subsequent reactor scram due to low reactor water level. The event was attributed to personnel error (i.e., failure to follow procedure). Corrective actions included counseling and remedial training for the individuals involved, as well as, the production and use of a training video tape on the event which was presented to each oncoming shift crew with emphasis placed on management's expectations with respect to self-checking.

Corrective actions for the previous LERs could not reasonably be expected to prevent the event documented by LER 97-001.

VI. Corrective Actions

The following corrective actions have been taken or are in progress:

1. The SRO, Operations Foreman, and Nonlicensed Operator were counseled concerning the event, expectations concerning pre-job briefings, and missed opportunities for preventing the event through self-checking and checking of others. On January 8, 1997, a Standing Instruction was issued for documented training by the operating crews on these same topics.

2. Operations personnel are developing a tool (e.g., checklist or flowchart) to aid the Operations Foreman in identifying high risk tasks and in determining the appropriate level of briefing detail. This tool will also help determine the extent of supervision needed based on the level of risk associated with the task.

3. Operations management is providing additional training to the operating crews on this event and on management expectations concerning communications and use of personnel error reduction techniques.

4. A multidisciplined work group is being formed to establish an Error Reduction program.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

The following table identifies those actions which are considered to be regulatory commitments. Any other actions discussed in this document represent intended or planned actions, are described for the NRC's information, and are not regulatory commitments. Please notify the Manager - Regulatory Affairs at the Perry Nuclear Power Plant of any questions regarding this document or any associated regulatory commitments.

Commitments
1. Operations personnel are developing a tool (e.g., checklist or flowchart) to aid the Operations Foreman in identifying high risk tasks and in determining the appropriate level of briefing detail. This tool will also help determine the extent of supervision needed based on the level of risk associated with the task. This tool will be developed and implemented by March 30, 1997.
2. Operations management is providing additional training to the operating crews on this event and on management expectations concerning communications and use of personnel error reduction techniques. This training will be completed by May 26, 1997.